10 CFR 50.73

CP&L

Carolina Power & Light Company
Robinson Nuclear Plant
3581 West Entrance Road
Hartsville SC 29550

Robinson File No: 13510C
Serial: RNP-RA/98-0170

SEP 23 1998

United States Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, DC 20555

H. B. ROBINSON STEAM ELECTRIC PLANT, UNIT NO. 2
DOCKET NO. 50-261/LICENSE NO. DPR-23

LICENSEE EVENT REPORT NO. 1997-011-01

Gentlemen:

The attached supplement to Licensee Event Report (LER) No. 97-011, originally submitted by letter dated December 16, 1997, is submitted in accordance with 10 CFR 50.73. The revised information is identified by a right hand margin bar. Should you have any questions regarding this matter, please contact Mr. T. M. Wilkerson, Manager, Regulatory Affairs.

Very truly yours,

J. W. Moyer
Director, Site Operations

PMY/py
Attachment
c: Mr. L. A. Reyes, NRC Region II
Mr. R. Subbaratnam, NRR
NRC Resident Inspector, HBRSEP

IE22

United States Nuclear Regulatory Commission
Attachment to Serial RNP-RA/98-0170

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NRC FORM 366 (6-1998) U.S. NUCLEAR REGULATORY COMMISSION

LICENSEE EVENT REPORT (LER)

(See reverse for required number of digits/characters for each block)

APPROVED BY OMB NO. 3150-0104 EXPIRES 06/30/2001

Estimated burden per response to comply with this mandatory information collection request: 50 hrs. Reported lessons learned are incorporated into the licensing process and fed back to industry. Forward comments regarding burden estimate to the Records Management Branch (T-6 F33), U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, and to the Paperwork Reduction Project (3150-0104), Office of Management and Budget, Washington, DC 20503. If an information collection does not display a currently valid OMB control number, the NRC may not conduct or sponsor, and a person is not required to respond to, the information collection.

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TITLE (4)
Reactor Trip Due to Condensate Pump "B" Shaft Failure

EVENT DATE (5)			LER NUMBER (6)			REPORT DATE (7)			OTHER FACILITIES INVOLVED (8)	
MONTH	DAY	YEAR	YEAR	SEQUENTIA	REVISION	MONTH	DAY	YEAR	FACILITY NAME	DOCKET NUMBER
11	16	1997	1997	011	01	09		1998		05000
1				--	--				FACILITY NAME	DOCKET NUMBER
										05000

OPERATIING MODE	1	THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR§: (Check one or more) (11)							
			20.2201(b)		20.2203(a)(2)(v)		50.73(a)(2)(i)		50.73(a)(2)(viii)
POWER			20.2203(a)(1)		20.2203(a)(3)(i)		50.73(a)(2)(ii)		50.73(a)(2)(x)
	100		20.2203(a)(2)(i)		20.2203(a)(3)(ii)		50.73(a)(2)(iii)		73.71
			20.2203(a)(2)(ii)		20.2203(a)(4)	X	50.73(a)(2)(iv)		OTHER
			20.2203(a)(2)(iii)		50.36(c)(1)		50.73(a)(2)(v)		Specify in Abstract below or in NRC Form 366A
			20.2203(a)(2)(iv)		50.36(c)(2)		50.73(a)(2)(vii)		

LICENSEE CONTACT FOR THIS LER (12)

NAME	TELEPHONE NUMBER (Include Area Code)
H.K. Chernoff, Supervisor , Licensing-Regulatory Programs	(843) 857-1437

COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT (13)

CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO EPIX		CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO EPIX
X	SG	P	B580	Y						

SUPPLEMENTAL REPORT EXPECTED (14)				EXPECTED	MONTH	DAY	YEAR
	YES (If yes, complete EXPECTED SUBMISSION DATE).	X	NO				

ABSTRACT (Limit to 1400 spaces, i.e., approximately 15 single-spaced typewritten lines) (16)

On November 16, 1997, at 0133 hours, with H. B. Robinson Steam Electric Plant, Unit No. 2 at one hundred percent (100%) power, an automatic actuation of the Reactor Protection System (RPS) occurred causing an automatic reactor trip. The RPS actuation resulted from "C" Steam Generator (SG) low level coincident with feedwater flow less than steam flow. The cause of the reactor trip was reduced feedwater flow to the SG following the failure of the "B" condensate pump stub shaft. The unit was stabilized in MODE 3 using the Emergency Operating Procedures. The NRC Operations Center was notified of this event at 0353 hours on November 16, 1997, in accordance with 10 CFR 50.72(b)(2)(ii). Plant equipment operated as designed with the exception of the intermediate range neutron monitoring system, channel N-35, which prevented the source range neutron monitoring channels from automatically re-energizing and providing indication.

A previous similar event occurred in 1991 when the "B" condensate pump stub shaft failed causing a reactor trip that was reported in LER 91-11. An investigation into the event reported in this LER has determined that the cause of the shaft failure was similar to the 1991 event. The cause of this event has been determined to be failure to adequately complete the corrective actions from the 1991 event. This report is submitted in accordance with 10 CFR 50.73(a)(2)(iv) as an event or condition that resulted in an automatic actuation of the reactor protection system.

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TEXT *(If more space is required, use additional copies of NRC Form 366A)* (17)

I. DESCRIPTION OF EVENT

On November 16, 1997, at approximately 0133 hours, with the H. B. Robinson Steam Electric Plant, Unit No. 2 (HBRSEP) operating at 100% power, plant operations personnel responded to a feedwater system (EIIS System Code: SJ) transient. The alarms and indications resulting from the feedwater transient caused the plant operators to enter Abnormal Operating Procedure (AOP)-010, "Main Feedwater/Condensate Malfunction," to determine the cause of, and response to, a loss of level in the steam generators (SG) (EIIS System Code: JB). Following the determination that no pumps were tripped in the feedwater system, and with level decreasing in the SGs, the operators initiated a manual runback of the turbine generator (EIIS System Codes: TA, TB) to reduce the steam flow and feedwater demand, and to attempt to manually control SG levels. The reactor protection system (RPS) (EIIS System Code: JC) automatically tripped the reactor (EIIS System Code: AA) as a result of low level in "C" SG coincident with feedwater flow less than steam flow. This anticipatory automatic reactor trip occurs whenever narrow range level in any one SG is less than 30% on 1 out of 2 SG level channels, coincident with feedwater flow less than steam flow by 6.4×10^5 lbm/hr on 1 out of 2 SG flow channels in the same SG.

The unit was stabilized in MODE 3, "Hot Standby," at the no load temperature of 547°F, at approximately 0150 hours using the Emergency Operating Procedures, Path 1, "Reactor Trip or Safety Injection," and EPP-4, "Reactor Trip Response." The Motor Driven Auxiliary Feedwater (AFW) Pumps (EIIS System Code: BA) automatically started on SG low-low level as required by the Engineered Safety Features (ESF) Actuation System (EIIS System Code: JE) to maintain secondary heat removal capability. The NRC Operations Center was notified of this event at 0353 hours on November 16, 1997 in accordance with 10 CFR 50.72(b)(2)(ii).

Plant equipment operated as designed with the exception of an intermediate range neutron monitoring channel, N-35, (EIIS System Code: IG) that was not properly compensated due to a failure of the compensation chamber and associated circuitry. The channel output and indication did not decrease below approximately 2×10^{-9} Ion Chamber Amps, which prevented the source range neutron monitoring channels from automatically re-energizing and providing indication. The source range channels were manually reenergized at approximately 0150 hours.

Prior to the event, the "B" condensate pump was in service delivering rated flow. During the event, operations personnel noted that the indication for the condensate pumps (EIIS System Code: SG; EIIS Component Function: P) and the feedwater pumps (EIIS System Code: SJ; EIIS Component Function: P) showed that the pumps were in operation. No structures, systems, or components were inoperable at the start of the event that contributed to the event.

TEXT *(If more space is required, use additional copies of NRC Form 366A)* (17)

II. CAUSE OF THE EVENT

An investigation into the cause of the feedwater transient determined that feedwater flow was reduced because the "B" condensate pump stub shaft failed. The preliminary investigation results have determined that this failure is similar to a failure of the "B" condensate pump stub shaft which occurred in 1991. That failure caused a reactor trip which was reported in LER 91-11. The cause of the shaft failure in 1991 was determined to be from cyclic fatigue due to the faulty design of the keyway. The preliminary investigation of the failure reported in this LER has determined that the stub shaft probably failed as a result of the same failure mechanism which caused the shaft failure in 1991. A failure analysis of the stub shaft is underway and will be completed by February 15, 1998. This analysis is expected to support the preliminary conclusions. The root cause of the event reported in this LER was determined to be inadequate implementation of corrective actions following the 1991 event.

The failure analysis of the stub shaft was completed and confirmed the preliminary conclusions that the stub shaft failed as a result of the same failure mechanism which caused the shaft failure in 1991.

HBRSEP, Unit No. 2 has two condensate pumps. Each condensate pump is a vertically-mounted, multistage pump. The pump body is mounted in a well. The pump driver is mounted on a platform above the pump body. The stub shaft connects the driver shaft to the pump shaft using pump couplings that incorporate a keyway design.

Following the 1991 failure, it was determined that a new design of the condensate pump shaft and stub shaft were needed. The pump manufacturer had redesigned the shafts to minimize the stress riser locations and increase the shaft cross-section in the vicinity of the shaft keyway. New stub shafts were ordered and the pump bodies were returned to the manufacturer to have new pump shafts installed. Work tickets were written to install the pump bodies with the redesigned shafts during Refueling Outage (RO)-14.

During the period of time between ordering the new stub shafts and pump shafts, and the stub shaft failure on November 16, 1997, both the "A" and "B" condensate pump bodies were replaced with refurbished pump bodies which incorporated the redesigned pump shafts. The "A" pump body replacement included a new design stub shaft. The "B" pump body replacement included a stub shaft of the old design which failed on November 16, 1997.

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TEXT *(If more space is required, use additional copies of NRC Form 366A)* (17)

III ANALYSIS OF THE EVENT

This report is submitted in accordance with 10 CFR 50.73(a)(2)(iv) as any event or condition that resulted in manual or automatic actuation of any engineered safety features, including the reactor protection system.

There was no adverse impact to safety as a result of this event. This event is bounded by the Updated Final Safety Analysis Report (UFSAR) Chapter 15 analysis for a loss of normal feedwater. The UFSAR analysis states that when the reactor trips due to the SG low-low level reactor trip, an adequate volume of inventory remains in the steam generators to assure a short-term controllable response. In this event, a RPS trip was received prior to the SG low-low level trip due to SG low level coincident with steam flow greater than feed flow. An alternate supply of feedwater from the safety related auxiliary feedwater system was available to assure cooling and orderly recovery of the unit.

IV. CORRECTIVE ACTIONS

The failed stub shaft was replaced with a redesigned stub shaft on November 19, 1997.

A failure analysis of the stub shaft was completed. The results of this analysis supported the engineering evaluation of the design adequacy of the new condensate pump shaft and stub shaft.

An engineering evaluation of the design adequacy of the new condensate pump shaft and stub shaft design was completed. The results of the evaluation determined future actions regarding design of the shaft and stub shaft were not required.

A review was completed for other long-shaft pumps for similar deficiencies including pumps that have been modified to add more stages. No design changes were determined to be needed by these reviews.

During performance of the "A" condensate pump preventative maintenance in RO-18, non-destructive testing was performed on the pump shaft and stub shaft ends to determine if the shafts were suitable for continued service. The testing concluded that there were no indications on the pump shaft or stub shaft ends and that continued usage was acceptable.

A review of selected equipment failures between 1991 through 1994 has been completed. The selected failures included those which led to plant transients that challenged continued on-line operation. The evaluation of the review results concluded that corrective actions to prevent recurrence were satisfactorily completed and that no additional actions were warranted.

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V. ADDITIONAL INFORMATION

Previous Similar Events

LER 91-11, Reactor Trip Due to Failure of Condensate Pump Shaft

This LER supplement provides the results of the corrective actions, including the reviews of selected equipment failures between 1991 through 1994 that were performed.

CATEGORY 1

REGULATORY INFORMATION DISTRIBUTION SYSTEM (RIDS)

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AUTH.NAME AUTHOR AFFILIATION
CHERNOFF,H.K. Carolina Power & Light Co.
MOYER,J.W. Carolina Power & Light Co.
RECIP.NAME RECIPIENT AFFILIATION

SUBJECT: LER 97-011-01:on 971116,automatic actuation of RPS occurred causing automatic reactor trip.Caused by reduced FW flow to SG following failure of "B" condensate pump stub shaft. Replaced failed sub shaft.With 980923 ltr.

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